NON-PUBLIC?: N
ACCESSION #: 9006220191
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 OF 6

DOCKET NUMBER: 05000410

TITLE: Manual Reactor Scram Due to Loss of Condenser Vacuum Caused by Instrument Air Line Break
EVENT DATE: 05/14/90 LER #: 90-009-00 REPORT DATE: 06/13/90

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 045

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Gary Thompson, System Support and Testing TELEPHONE: (315) 349-2708

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: LD COMPONENT: PSP MANUFACTURER: X999
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On May 14, 1990, at approximately 2052 hours, with the reactor mode switch in the "RUN" position and the reactor operating at 100% rated thermal power, Nine Mile Point Unit 2 (NMP2) was experiencing numerous off-normal plant conditions. Operations personnel were responding to several abnormal Offgas System (OFG) indications and alarms, and Main Condenser vacuum was decreasing. At approximately 2058 hours, operators initiated reactor power reduction in response to decreasing condenser vacuum. At 2119 hours with reactor power at approximately 45% rated thermal power, the reactor mode switch was placed in the "SHUTDOWN" position, initiating a reactor scram.

The cause of this event was the partial loss of the Instrument Air System (IAS) due to pipe failure induced by stress-corrosion cracking.

Immediate corrective actions included: performing a walkdown of the suspected branch of the IAS to locate source of air loss; generating a Work Request to replace the failed air line; and requesting a failure analysis be performed to identify cause of pipe failure.

END OF ABSTRACT

I. DESCRIPTION OF EVENT

On May 14, 1990, at approximately 2052 hours, with the reactor mode switch in the "RUN" position and the reactor operating at 100% rated thermal power, Nine Mile Point Unit 2 (NMP2) was experiencing numerous off-normal plant conditions. Several Offgas System (OFG) indications and alarms were being responded to by Operations personnel including:

- Erratic OFG Recombiner Train A and B flowrate.

- High Offgas System inlet pressure.

- Offgas Condensers A and B low water level alarms.

- Abnormal indications (closed) on the pressure control valves supplying auxiliary steam to the OFG Preheaters.

- High pressure alarms on the auxiliary steam supply header to the OFG preheaters.

While responding to Offgas System alarms, operators noted a steady decrease in Main condenser vacuum, and that Service Water System Air Operated Valve 2SWP-HV98A (Condenser Air Removal Pump Seal Water Cooler Discharge Valve) failed open (valve position indication changed from green to red). At this point, operators suspected an instrument air failure. Control Room indications for the Instrument Air System were checked and found to be satisfactory (normal air pressure). This indicated that a localized loss of instrument air was occurring as opposed to a total loss of instrument air. In response to this, operators were dispatched to the Turbine Building to search for the source of the air leak.

Close attention was being paid to condenser vacuum status as operators were aware that the Offgas System's performance (control valves effecting condenser vacuum) was severely impaired by the IAS problems. At approximately 2058 hours, the Station Shift Supervisor (SSS) initiated reactor power reduction in response to the decreasing Main Condenser vacuum in accordance with Operating Procedures N2-OP-9, "Condenser Air Removal", and N2-OP-101D, "Power Changes". Reactor recirculation flow was reduced to minimum and twelve control rods with high rod worth were inserted.

At 2119 hours, with reactor power reduced to approximately 45% rated thermal power (52 percent core flow) and condenser vacuum reduced to 22.9 inches mercury (Hg.), Operations personnel initiated a reactor scram by placing the reactor mode switch in the "SHUTDOWN" position. Operators then proceeded with scram recovery per Operating Procedure N2-OP-101C, "Plant Shutdown".

Reactor water level momentarily dropped to 145 inches due to the shrink caused by the rapid power reduction. Operators immediately entered Emergency Operating Procedure N2-EOP-RPV, "Reactor Pressure Vessel Control-Flowchart", (required actions when reactor water level is less than 159.3 inches). Normal reactor water level was then restored by manual control of the feedwater system.

At 2120 hours, the Main Turbine Generator tripped on Main Generator anti-motoring trip. House loads automatically transferred to offsite power.

At 2123 hours, operators reset the reactor scram.

At approximately 2125 hours, Emergency Operating Procedure N2-EOP-RPV was exited as reactor water level was restored to normal band (178 to 187 inches).

During the event, there were no unexpected plant system responses. There were no Emergency Core Cooling System (ECCS) actuations.

Following the event, a visual inspection of the suspected Instrument Air System branch identified fractured piping in the Low Conductivity Waste Tank Room. The damage to the piping consisted of two longitudinal fractures approximately five feet long, 180 degrees apart on the same horizontal piece of pipe, and a third longitudinal fracture, approximately one foot long, at one end of this pipe section. Piping material is two inch schedule 40 red brass ASTM B43 "Specification For Seamless Red Brass Pipe, Standard Sizes" and is operated at a pressure of approximately 110 pounds per square inch gauge (psig).

II. CAUSE OF EVENT

The immediate cause for this event was a loss of vacuum to the Main Condenser. Sequential causes for this initiation were:

- Loss of instrument air due to a ruptured Instrument Air Line (2IAS-002-420-4) downstream of two inch air line

header (2IAS-002-411-4) used to supply air to various Turbine Building loads.

- This break caused Instrument Air System Excess Flow Check Valve 2IAS-EFV596 to close, isolating air to the Offgas System air operated control valves.

- The loss of instrument air and the effect this had on Offgas System valves caused Main Condenser vacuum to decrease.

Failure analysis examinations were conducted and completed on portions of the damaged two inch brass piping to determine the cause of the fractures. It was concluded that the probable cause for the pipe failure was ammonia initiated stress-corrosion cracking.

Brinell hardness test results indicated that the piping was in a cold worked condition. Additionally, ammonia, in the presence of oxygen and water vapor, and placed in contact with stressed copper alloys, potentially can create a phenomenon known as "season cracking" (a form of stress-corrosion cracking) in the copper alloy materials.

Ammonia vapor has been known to be present in the room where this piping is installed as a result of the regeneration of deionizing resins during NMP2 initial startup testing, and during introduction of new resins into the system. The tanks used for regeneration and processing of new resins are located within the Low Conductivity Waste Tank Room, and vent to atmosphere within the room. Regeneration of the resin in this location was abated approximately two years ago; however, new resins are introduced into the system periodically, which can produce ammonia based vapor.

III. ANALYSIS OF EVENT

This event is reportable in accordance with 10CFR50.73 (a) (2) (iv):

"Any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS). However, actuation of an ESF, including RPS, that results from and was part of the preplanned sequence during testing or reactor operation need not be reported".

The manual reactor scram initiation by Operations personnel was due to the placing of the reactor mode switch in the "SHUTDOWN" position. This response action was the result of a decrease in Main Condenser vacuum below 23 inches Hg.

There were no Emergency Core Cooling System (ECCS) actuations during this event. This event was significantly less severe than the bounding accident analysis in the NMP2 Updated Safety Analysis Report, Chapter 15.

A manual reactor scram is a protective function and a conservative event, and poses no adverse safety consequences at any power level. This event did not adversely affect any safety system nor the operators ability to achieve safe shutdown.

IV. CORRECTIVE ACTIONS

Immediate corrective actions included:

A. Performing an extensive visual inspection of the suspected branch of the IAS piping to locate air leak.

B. Initiating a Work Request (WR 176538) to remove and replace the failed IAS line.

C. Requesting a failure analysis be performed on the ruptured piping to determine cause of the fracture.

As a result of the findings from the failure analysis, all one inch and two inch red brass IAS piping located in the Low Conductivity Waste Tank Room was removed and replaced in kind per WR 169907. This replacement decision was based upon the following information:

A. Ammonia vapor is no longer produced due to regeneration of existing resins. This process is considered the significant contributor of ammonia vapor in the Low Conductivity Waste Tank Room.

B. Ammonia vapor that is produced in this room during infrequent resin addition, is of less significant concentration. Based upon this condition, the piping failure analysis, and IAS System service history, replacement piping service life is certain to be much greater than the duration of the current unit operating cycle.

Long term corrective action includes performing a review/determination as to whether the present pipe material located within the Low Conductivity Waste Tank Room should be replaced during the first refueling outage by material not suspectable to ammonia related stress corrosion.

V. ADDITIONAL INFORMATION

A. Previous similar events: none.

B. Failed component identification:

2" schedule 40 red brass piping
ASTM B43 "Specification For Seamless Red
Brass Pipe, Standard Sizes".

C. Identification of components referred to in this LER:

IEEE 803 IEEE 805
COMPONENT EIIS FUNCT. SYSTEM ID

Service Water System N/A BI
Offgas System N/A WF
Instrument Air System N/A LD
Control Valves (HV) HCV BI
Main Condenser COND SG
Offgas Condenser COND WF
Offgas Recombiner RCB WF
Pressure Control Valves (OFG) PCV WF
Offgas Preheaters HX WF
Excess Flow Check Valve ISV LD
Turbine Generator TG TA
Control Room N/A NA
Turbine Building N/A NM

ATTACHMENT 1 TO 9006220191 PAGE 1 OF 1

NIAGARA MOHAWK

NINE MILE POINT NUCLEAR STATION/P. O. BOX 32, LYCOMING, N.Y. 13093
TELEPHONE (315) 343-2110

NMP 67567

June 13, 1990

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 90-09

Gentlemen:

In accordance with 10CFR50.73, we hereby submit the following Licensee Event Report:

LER 90-09 Is being submitted in accordance with 10CFR50.73 (a) (2) (iv), "Any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)".

A 10CFR50.72 (b) (2) (ii) report was made at 2255 hours on May 14, 1990.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

J. L. Willis
General Superintendent
Nuclear Generation

JLW/GB/lmc

ATTACHMENT

xc: Regional Administrator, Region I
Sr. Resident Inspector, W. A. Cook

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